NUREG/CP-0174

Transactions of the Twenty-Ninth Nuclear Safety Research Conference

(formerly, The Water Reactor Safety Information Meeting)

To Be Held at
Marriott Hotel at Metro Center
Washington, DC
October 22-24, 2001

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

Proceedings prepared by
Brookhaven National Laboratory

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at www.nrc.gov/NRC/ADAMS/index.html.
Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer, Reproduction and Distribution Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address www.nrc.gov/NRC/NUREGS/indexnum.html are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—
The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—
American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

DISCLAIMER: Where the papers in these proceedings have been authored by contractors of the U. S. Government, neither the U.S. Government nor any agency thereof, nor any U.S. employee makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use or the results of such use, of any information, apparatus, product, or process disclosed in these proceedings, or represents that its use by such third party would not infringe privately owned rights. The views expressed in these proceedings are not necessarily those of the U. S. Regulatory Commission.

NUREG/CP-0174

Transactions of the Twenty-Ninth Nuclear Safety Research Conference

(formerly, The Water Reactor Safety Information Meeting)

To Be Held at Marriott Hotel at Metro Center Washington, DC October 22-24, 2001

Date Published: October 2001

Compiled by: Susan Monteleone, Meeting Coordinator

S. Nesmith, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

PREFACE

This report contains summaries of papers on reactor safety research to be presented at the 29th Nuclear Safety Research Conference (formerly known as the Water Reactor Safety Information Meeting) at the Marriott Hotel at Metro Center in Washington, DC, October 22-24, 2001. They briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Also included are summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry. The abstracts have been compiled here to provide a basis for meaningful discussion of information exchanged during the course of the meeting, and are in the order of their presentation on each day of the meeting.

This page left blank intentionally

Contents

Page

29th Nuclear Safety Research Conference
October 22-24, 2001
Washington, D.C.

Monday, October 22, 2001

TECHNICAL SESSION 1: ADVANCED REACTORS
Chaired by: T. King (NRC), S. Floyd (NEI)

Objective: This session will cover the specific research work needed to prepare the NRC for reviewing the design of advanced reactors.

TECHNICAL SESSION 2: DRY CASK STORAGE RESEARCH
Chaired by: M. Cunningham (NRC), J. Guttman (NRC)

Objective: This session will detail recent accomplishments of the Office of Research in assessing important safety and risk issues related to licensing dry-cask fuel storage devices. Plans for future studies will be discussed.

Tuesday, October 23, 2001

TECHNICAL SESSION 3: AGE-RELATED ISSUES AND RESEARCH
Chaired by: N. Chokshi (NRC), W. Bateman (NRC)

Objective: This session will present the most recent data from NRC's research on aging and age-related degradation in nuclear power plants, and how this relates to current operating experiences and other emerging issues. Also discussed will be the relevance of information on aging management from other industries.

TECHNICAL SESSION 4: FUEL RESEARCH
Chaired by: R. Meyer (NRC), R. Yang (EPRI)

Objective: This session will encompass the recent accomplishments and results from the NRC's fuel research program and their related international cooperative agreements.

Wednesday, October 24, 2001

TECHNICAL SESSION 4: FUEL RESEARCH (Cont'd)
Chaired by: R. Meyer (NRC), R. Yang (EPRI)

TECHNICAL SESSION 5: RISK INFORMING REGULATORY PRACTICES
Chaired by: M. Cunningham (NRC), C. Carpenter (NRC)

Objective: This session will communicate recent accomplishments and future plans in the Office's work to incorporate risk information into NRC's regulatory practices.

This page left blank intentionally

ADVANCED REACTOR GROUP (ARG) PRE-APPLICATION REVIEW ACTIVITIES

John Flack, Chief
Regulatory Effectiveness Assessment & Human Factors Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Pre-application interactions with potential licensee applicants help prepare both the NRC and nuclear industry for future licensing reviews. The objectives of these reviews include the development of a licensing approach and guidance necessary to enable the regulatory process to address nuclear designs that substantially deviate from current generation reactors. Pre-application reviews raise and provide insight into design, safety, licensing and policy issues that would need to be resolved as part of the licensing process. In addition, pre-application reviews help to develop the infrastructure that is necessary to independently assess the safety capacity of advanced reactor designs, including the development of analytical tools, testing, and experiments that would be necessary to validate methods and tools. This presentation will cover the purpose, status, and up-to-date insights that stem from of the ARG pre-application review activities. These activities include the Exelon's Pebble Bed Modular Reactor, and other soon to be submitted advanced reactor designs such as General Atomics' Gas Turbine-Modular Helium Reactor (GT-MHR) and Westinghouse International Reactor Innovative and Secure (IRIS) design.

This page left blank intentionally

NUCLEAR REGULATORY COMMISSION
PREPARATION FOR FUTURE REACTOR LICENSING

James E. Lyons, Director
US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
New Reactor Licensing Project Office

In response to a renewed interest in building nuclear power plants, the NRC has created organizations within its major program offices to prepare the NRC staff for new applications (e.g., early site permits (ESPs), design certifications, and combined licenses) and to manage special task groups and pre-application reviews of new reactor designs. Activities include (1) evaluating the ability of the NRC staff to support future application reviews under 10 CFR Parts 50 and 52; (2) performing ESP reviews and pre-application reviews of the AP1000 (a light-water reactor design with passive safety systems), Pebble Bed Modular Reactor (a high-temperature gas-cooled reactor design), International Reactor Innovative and Secure (an advanced light-water reactor design), and Gas Turbine-Modular Helium Reactor (a high-temperature gas-cooled reactor design); (3) initiating and/or performing related rulemakings that will update 10 CFR Part 52 to reflect lessons learned from certifying three nuclear plant designs, update Tables S-3 and S-4 of 10 CFR Part 51 to address higher burnup fuel considerations and non-LWR advanced designs, and address alternative siting considerations; (4) reactivating the construction inspection program, and (5) interacting with stakeholders to ensure there is a clear understanding of upcoming activities related to future applications and to solicit stakeholder input.

Future activities are expected to include (1) managing the reviews of five new applications resulting from the pre-application reviews (including one design certification, one combined license, and three ESP reviews), (2) managing two pre-application reviews (IRIS and GT-MHR), (3) updating regulatory and review guidance for new applications, i.e., Standard Review Plans (SRPs), Regulatory Guides, and referenced codes and standards, and identifying where enhancements are needed, (4) developing independent codes to analyze the safety of non-LWR designs, with supporting validation testing, and (5) addressing regulatory infrastructure issues, including a proposal by the Nuclear Energy Institute on a generic regulatory framework and NRC regulations governing financial issues and operator staffing.

This page left blank intentionally

A NEW RISK-INFORMED DESIGN AND REGULATORY PROCESS

George Apostolakis and Michael Golay
Massachusetts Institute of Technology

Allan Camp and Felicia Duran
Sandia National Laboratory

David Finnicum and Stanley Ritterbusch
Westinghouse Electric Company

In a project funded by the USDOE in its Nuclear Energy Research Initiative Program the authors have been involved in formulating a new risk-informed nuclear safety regulatory approach that we hope will be sufficiently attractive that it can be adopted for use by the NRC.

We believe that this work is important because a new regulatory treatment is needed both for the licensing of new non-light water reactors (LWRs), and to rationalize the regulation of LWRs. It is common today for the plans for new reactor concepts to include proposals for how they should be licensed. The existence of such proposal is implicit evidence that the existing regulatory structure is inadequate for this purpose. Similarly attempts to "risk inform" the regulations governing LWRs have made only small progress because of the complexity and inconsistency of the existing structure. Thus, we have concluded that a fresh start in formulating a regulatory structure is worth attempting. This paper describes the fundamental concepts of that attempt.

The overall purpose of the new approach, termed Risk-Informed Regulation, is to formulate a method of regulation that is logically consistent and devised for both the reactor designer and regulatory to work together in obtaining systems able to produce economical electricity safely. In this system the traditional tools (deterministic and probabilistic analyses, tests and expert judgement) and treatments (defense-in-depth, conservatism) of safety regulation would still be employed, but the logic governing their use would be reversed from the current treatment. In the new treatment probabilistic risk analysis (PRA) would be used as the paramount decision support tool, taking advantage of its ability to integrate all of the elements of system performance and to represent the reflects of uncertainties in these results. The latter is the most important reason for this choice, as the difficult part of safety regulation is the treatment of uncertainties, not the assurance of expected performance.

The scope of the PRA would be made as large as that of the reactor system, including all of its performance phenomena. The models and data of the PRA would be supported by deterministic analytical results, and data to the extent feasible. However, inevitably these would require being complemented by subjective judgements where the former was inadequate (to doing this is always necessary if decisions are to be made). All of these elements play important roles in the current decision-making structure; the main departure from current practice would be making all of these treatments explicit within the PRA.

In the intended sense the PRA would be used as a vehicle for stating the beliefs of the regulatory decision-maker; the foundation of his decisions. Thus, the PRA should be viewed as a Bayesian decision tool, and be used in order to take advantage of its capabilities in integration and inclusion of uncertainties. In order to do this all regulations must be formulated in terms of acceptable levels of unavailability of essential functions, including an acceptable level of uncertainty (e.g., the acceptability of system performance could be evaluated at a stated confidence level rather than in terms of the mean).

Implied in this treatment is a hierarchy of acceptable performance goals. At the highest level societal Safety Goals would be used, supported by subgoals formulated at increasingly fine levels of detail as the hierarchical level of the goal would decrease.

In the proposed treatment the use of defense-in depth and requiring performance margins would remain as they do currently. However, the current practice of permitting such features to be required without justification would be abandoned; rather, wherever such a requirement were to be made, it would also be necessary for the regulator to provide evidence concerning the value of the requirement and to reflect that value in the master PRA (i.e., if a redundancy is to be worth including in a system its safety value should also be stated in the overall system performance analysis).

Much work remains to be done in developing this regulatory treatment to the point that it can be employed in practical decision-making. Questions of how to set subordinate performance goals, of quality standards for PRA models and data, of procedures for incorporating subjective judgements into parse reliably and of how to formulate regulatory decision rules based upon PRA results must all be addressed. In order to do this in a fashion able to support the licensing of advanced reactors it is necessary for the DOE and the NRC to cooperate in a multi- year campaign of regulatory research. At the moment this collaboration has not begun, but it is vital to the future success of efforts to provide the nation with improved nuclear power options.

The Need for a New Regulatory Framework

Stephen D. Floyd, Senior Director – Regulatory Reform
Nuclear Energy Institute

Today's reactor regulatory process is based on the same concepts and principles as it was 35 years ago: deterministic design-basis events. The current regulations have provided for an adequate level of protection of public health and safety. Yet, operating experience and risk analyses insights have revealed that the process could be significantly enhanced by increasing regulatory focus and attention on some requirements, while other requirements could be significantly reduced or eliminated. The adoption of a complete risk-informed, performance-based approach would significantly enhance the protection of public health and safety through increased licensee and NRC attention and focus on safety significant matters.

In a competitive generating market, plant safety must continue to be of paramount importance. Risk analyses and monitoring tools and processes are available that would allow the NRC to provide licensees additional flexibility in the manner in which they can implement the regulations, while at the same time improving the protection of public health and safety. A risk-informed, performance-based process will allow licensees to implement, and NRC staff to oversee, the regulations in a more efficient and effective manner, which will improve safety. Failure to accomplish these needs could result in obsolescence, unnecessary plant closures and slow erosion in safety performance as innovation and potential safety enhancements are asphyxiated by an unnecessarily prescriptive and rigid regulatory process.

The presentation will describe the overall framework and regulatory approach to develop a risk-informed, performance-based regulatory process for new power reactors.

This page left blank intentionally

HTGR RESEARCH PROGRAMS AND NEEDS

Thomas L. King, Director
Division of Systems Analysis & Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Current research activities and additional research needs related to the high temperature gas-cooled reactor (HTGR) design will be discussed. HTGRs present unique safety issues to be resolved and policy issues for Commission consideration. As a result of renewed interest in licensing HTGRs in the U.S., NRC hosted an international workshop to discuss safety issues, their research needs and priorities. Results from this workshop will be discussed.

HTGR research programs encompass three broad areas; new and different fuel designs, materials and structures, and safety analyses. The fuel research includes fuel characteristics, issues with fabrication, fuel behavior during steady and transient conditions, and testing required for fuel qualification. Research to provide better understanding of the source term is also planned. Materials and structures research includes the study of characteristics of metal, graphite and concrete under HTGR-specific thermal and irradiated conditions. Safety analysis research consists of identification of HTRG-specific design features; tests to confirm thermal hydraulic and neutronic phenomena of a HTGR; and the development, verification and validation of analytical tools to predict steady state and transient behavior of the reactor and safety systems.

This page left blank intentionally

Industry Views on R&D Needs for New Plants

Edward Rodwell, Manager
Advanced Nuclear Plants' Systems
Electric Power Research Institute

Results of the nuclear power section of a 1998 comprehensive EPRI-led industry review of key US electricity generation R&D needs are summarized. A resulting and prompt redirection, of modestly expanding resources available for future nuclear plant R&D, is described, to new projects that will contribute to the availability of near-term nuclear plant deployment options with enhanced economic competitiveness. Each project is described as addressing cost-reduction via plant design, or cost-reduction via construction or operation technology, or demonstration of a specific feature of a cost-reducing design, or enhancement of deployment readiness. Both light water reactor and helium gas-turbine reactor technologies are addressed, because of their potential for economically competitive near-term deployment. Achievements by these projects over the last three years are described, along with an estimate of the resulting improvements to the economic competitiveness of near-term nuclear plant deployment options and to their readiness for deployment. The planned remaining scope of these projects is described. Needed R&D identified but not yet funded and initiated (mainly in the arenas specific feature demonstration and of deployment readiness enhancement) is also noted.

This page left blank intentionally

Pre-application Activities for IRIS and Related Research Needs

by

M. D. Carelli and S. E. Ritterbusch
Westinghouse Electric Co.

IRIS (International Reactor Innovative and Secure) is a new generation advanced light water reactor being developed by an international consortium led by Westinghouse. This consortium of 18 organizations from nine countries has completed the conceptual design and a limited pre-application licensing activity is planned in FY02 as the first step towards the goal of achieving US Nuclear Regulatory Commission design certification by 2008.

Major innovative characteristics of IRIS are: integral design, where all primary system components (pumps, steam generators, pressurizer) are enclosed within the reactor vessel; straight burn of a five-year-life core with less than 5% enriched fuel; at least 48 months interval for maintenance shutdowns; and safety- by-design approach whereby design postulated accidents are eliminated or their probability of occurring is lessened and consequences reduced. Another unique IRIS feature is the thermohydraulic coupling of the vessel with a small spherical high pressure containment which keeps the core covered for extended periods of time during small to medium LOCAs. This, coupled with the integral design that eliminates large LOCAs, allows IRIS to dispense with traditional emergency core cooling systems.

NRC Research needs related to IRIS are expected to fall under three major areas:

1. IRIS is a simplified design, where many traditional LWR accident sequences, starting with LOCAs, are not credible. Also not applicable are shorter than 48 months' maintenance-related requirements. Existing regulations need to be reviewed for their application to IRIS and a substantial streamlining is expected.

2. IRIS is based on proven LWR technology and thus a "prototype" is not needed for design certification. However, its innovative engineering must be tested to prove its predicted performance. A series of out-of-pile tests, both of individual components and of their integral behavior, is planned. These tests will be conducted on properly scaled models. Existing facilities will be used to the maximum extent possible, taking advantage of the worldwide distribution of the IRIS team members.

3. One of the stated goals by DOE for the Generation IV reactors is the elimination of the need for off-site emergency response. The safety-by-design approach that exploits the integral configuration characteristics provides a very strong deterministic basis. A preliminary analysis has indicated that of the ANSI 18.2 Class IV accidents considered in a typical PWR SAR, only one remains in IRIS, with much lower probability. All the others are either eliminated or downgraded to Class III or lower. In parallel, risk informed regulation will be used in the IRIS licensing process. Application of risk informed regulation to current LWRs has

indicated that the accident sequence probability can be substantially reduced. A comprehensive use of probabilistic methods, including modeling of uncertainties, is expected to provide a quantitative decision making tool in the IRIS design and licensing process. It is possible, therefore, that the combination of IRIS design characteristics with a risk informed process might result in an overall release probability justifying no need for off-site emergency response.

Westinghouse intends to engage NRC staff from the very beginning in the development process for the three areas above, by reviewing, commenting and critiquing IRIS plans. The top priority is represented by area 2, which is planned to start in the second quarter of FY02. Any slippage in the safety tests will result in a one-to-one delay of the overall schedule. The first activity will be the planning of the tests and preparation for each test of its similitude requirements and analyses.

THE PACKAGE PERFORMANCE STUDY: A STUDY OF SPENT FUEL TRANSPORTATION

Andrew J. Murphy and Robert J. Lewis
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555, USA 301-415-6011

Jeremy L. Sprung and Ken Sorenson
Sandia National Laboratories
Albuquerque, NM, 87185, USA 505-844-0134

ABSTRACT

The U.S. Nuclear Regulatory Commission's (NRC's) responsibilities in the transport of spent nuclear fuel include certification of transport packaging designs, approval of transport package Quality Assurance programs, issuance of general licenses authorizing licensees to offer material to carriers for transport, and establishment of physical protection requirements for spent fuel in transit. The Commission has been studying safety in the transport of spent nuclear fuel under its regulations for nearly 25 years. In December 1977, when the Commission accepted the generic environmental impact statement for transportation, it directed that regulatory policy concerning transportation be subject to close and continuing review.

The spent fuel transportation Package Performance Study (PPS) investigates the performance of casks and behavior of fuel when subjected to thermal and impact forces that exceed the hypothetical accident conditions specified in 10 CFR Part 71. Issues related to the probability and consequences of severe transport accidents will be examined. The objective of the PPS is to verify analytical models used to predict accident risk associated with transportation of spent fuel in NRC certified casks by comparing model predictions to the results of impact and fire tests. The PPS is a follow-on project to NUREG/CR-6672, "Reexamination of Spent Fuel Shipment Risk Estimates," which was published in March 2000. An enhanced public participatory process has been used in developing the PPS issues for study and the testing and analysis plans; this enhanced public participation will continue through the study.

As an introduction to the PPS, this paper will first review the studies of the safety of the transportation that the NRC has conducted over the last three decades and summarize the results obtained. Then, the paper will outline the objectives and scope of the PPS testing and analysis program.

This page left blank intentionally

A Risk Analysis of Spent Nuclear Fuel in Dry Casks

Christopher Ryder, Edward Rodrick, Jack Guttmann
U.S. Nuclear Regulatory Commission

The spent fuel pools of commercial nuclear power plants are becoming filled while the search for a permanent repository continues. Many utilities have been removing fuel from the pools and storing it in dry casks on site. The NRC Office of Nuclear Materials Safety and Safeguards, which licenses these casks, wants to quantify the risk of dry storage of spent nuclear fuel. This quantification will be used for considering options for risk-informing 10 CFR 72 regulatory requirements (including inspection programs), considering various options for safety goal development, enhancing public confidence, increasing regulatory efficiency and effectiveness, reducing unnecessary regulatory burden, and assessing the extent to which the collection of data on the performance of the casks in the field needs to be improved.

The NRC Office of Nuclear Regulatory Research is performing a pilot PRA of a spent fuel dry cask storage system, the Holtec International HI-STORM 100. This cask is being studied at a specific BWR site where the operations can be observed and modeled. (Although developed for a specific cask at a specific site, the analytical models developed for this preliminary study can be modified and applied to other dry cask systems at other reactor sites.) During its service life, the casks have three operational modes—handling in the reactor building, transfer to the storage pad, and storage for 20 years. In each of these modes, accidents that could result in mechanical and thermal challenges to the cask and that have the potential to cause the release of radioactive material, are postulated. Event tree/fault tree methods are used to develop logic models of plausible accident sequences. Engineering analyses are used to determine the stresses that would be imposed by the postulated events. Fracture mechanics are used to determine the probability of a cask failing when subjected to the stresses from postulated accident conditions. A human reliability analysis is used to determine the probability of accidents caused by human errors during cask operations, such as when the cask is moved while inside the reactor building or while being monitored during storage.

The preliminary results of the PRA suggest that the risk of the HI-STORM cask at the BWR plant is low compared to the risk of accidents involving the core of operating nuclear power plants. Events that have a high conditional probability of failing the cask have a low frequency (on the order of 10^{-6} per year or less). Conversely, events that occur with a high frequency have a low conditional probability (on the order of 10^{-6} or less) of failing the cask. Furthermore, the consequences of most events that have been postulated to fracture the cask and the fuels are low because the energy driving the radionuclides from the fuel pellets is low and the inventory of radionuclides in the fuel pellets is relatively low compared to the reactor inventory. Accordingly, the risk appears to be low.

This page left blank intentionally

Seismic Behavior of Spent Fuel Storage Cask Systems

S. Khalid Shaukat
U.S. Nuclear Regulatory Commission

The U. S. Nuclear Regulatory Commission (NRC) is conducting a research program to investigate technical issues concerning the dry storage of spent nuclear fuel by conducting confirmatory research for establishing seismic criteria and review guidelines for the seismic behavior of these systems. The program focuses on developing finite element analysis models that address the dynamic coupling of a module/cask, a flexible concrete pad, and an underlying soil/rock foundation, in particular, the soil-structure-interaction. Parametric analyses of the coupled models are planned to include variations in module/cask geometry, site seismicity, foundation properties, and cask/pad interface friction. The analyses performed and planned include:

1) a rectangular dry cask module typical of Transnuclear West design at a site in Western USA which typically has high seismicity;

2) a cylindrical dry cask typical of Holtec design at a site in Eastern USA which typically has low seismicity; and

3) a cylindrical dry cask typical of Holtec design at a site in Western USA with medium to high seismicity.

The paper includes assumptions made in seismic analyses performed to date, results, and conclusions.

Status and Results:

Seismic analyses of rectangular Transnuclear West Cask design at a site in Western USA were performed. The analyses were conducted using an artificial time history based on R.G. 1.60 (1.5g in each horizontal direction, and 1.0g in vertical direction) and also using time history for actual Tabas earthquake records (1.5g in each horizontal direction, 1.0g in vertical direction). These analyses were 3-D coupled nonlinear finite element analyses with cask, pad and soil underneath. A range of coefficient of friction was applied. The worst coefficient of friction assumed was 0.3, although it is extremely conservative and not realistic for concrete to concrete friction which is believed to be higher than 0.3. The conclusion was that the 3-cask module tied together may slide but the horizontal displacement will be less than half the clear distance between the neighboring modules and the edge distance of the pad. The cask will not slide off the pad and will not tip over.

Similar seismic analyses of cylindrical casks at a site in Eastern USA were performed using a 3-D coupled nonlinear finite element model of the cask, pad and soil underneath. The conclusion is that the cask will slide less than 0.1" assuming coefficient of friction 0.25, and seismic excitation level of 0.15g. Tip over will not occur.

Similar but a little more complicated seismic analysis of cylindrical casks at a site in Western USA is expected to be performed by the end of FY 2001. The complication is due to an additional layer of soil cement between the pad and the soil underneath. This introduces an additional variable being the coefficient of friction between pad and cement, and between cement and soil. The ground motion to be used is the one provided by the cask vendor (0.728g in one horizontal direction, 0.707g in the other horizontal direction, and 0.721g in vertical direction).

Work to be done:

Some additional work under consideration is the sensitivity analysis in the seismic behavior of the same cylindrical dry cask by varying the following:

1) Friction coefficient between cask and pad

2) Friction coefficient between pad and cement layer

3) Friction coefficient between cement layer and soil properties

4) Use of a different time history (e.g., actual Taiwan earthquake)

A review panel of eight members (four sponsored by the NRC, and four sponsored by EPRI) was established at the beginning of this project to provide direction and guidelines on how the work should proceed. The four sponsored by the NRC include three staff members and a nationally recognized expert. The four sponsored by EPRI include one from EPRI, and one nationally recognized expert and two representatives from the Industry. A review panel meeting is expected after completing the above-mentioned analyses. All the review panel members will be provided the information developed to seek guidance on the progress and direction of the work on generic applicability of these analyses. Generic review guidelines will be developed by evaluating a large number of cases with varying friction coefficients, soil properties, and seismic excitation levels. The end product is expected to be in the form of nomograms, charts, and/or tables that could be used as a tool for NRC review of licensee applications.

The schedule for completion of these efforts is November 2002.

An Analysis of a Spent Fuel Transportation Cask
Under Rail Tunnel Fire Conditions

Chris Bajwa
U.S. Nuclear Regulatory Commission

Hazards in the transportation industry from rail shipments can result in accidents involving fire. Fire is a design basis event for certifying casks used in transportation of radioactive material. Title 10 of the Code of Federal Regulations Part 71 section 73(c)(4), (10 CFR 71.73(c)(4)) delineates that transportation packages designed to ship radioactive material and that are subject to accident conditions must be designed to resist a devouring fire of a duration of 30 minutes in order to protect the contents of the package and prevent release of radioactive material to the environment.

Recently, a fire occurred in a railroad tunnel outside of Baltimore Maryland, involving hazardous materials. According to reports, the fire burned for several days. Although it is very unlikely that a devouring fire of such duration could occur during the shipment of spent nuclear fuel, several questions have been raised with regards to how a spent fuel cask would perform if exposed to a fire inside a tunnel for several days.

The staff of the Spent Fuel Project Office (SFPO), in the Office of Nuclear Material Safety and Safeguards (NMSS) is responsible for evaluating the performance of spent fuel transportation casks under the fire accident conditions specified in the regulations. Evaluating the thermal performance of a transportation package design is essential to ensuring that a package can survive a fire accident and will not release radioactive material in excess of the release limits established by the NRC.

This paper will examine a spent fuel transportation cask with a welded canister under conditions similar to the severe fire that occurred in the Baltimore, Maryland railroad tunnel. The ANSYS finite element analysis program will be utilized for this analysis. The paper will describe the model that was utilized for the analysis and present preliminary results, as well as a discussion on the significance of the results.

This page left blank intentionally

Stress Corrosion Cracking and Non-Destructive Examination of Dissimilar Metal Welds and Alloy 600

Deborah A. Jackson
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

The United States Nuclear Regulatory Commission (USNRC) has conducted research in the areas of assessment and reliability of Non-Destructive Examination (NDE) and environmentally assisted cracking since the 1977. Since that time work has continued but recent occurrences of cracking in Inconel (Alloy 82/182) welds and Alloy 600 base metal at several domestic and overseas plants have raised several issues. The occurrences of cracking have been identified through indirect means, specifically the discovery of boric acid deposits resulting from through-wall cracking in the primary system pressure boundary. Analyses indicate that the cracking has occurred due to primary water stress corrosion cracking (PWSCC) in Alloy 82/182 welds, in both hot leg nozzle-to-safe end welds and control rod drive mechanism (CRDM) nozzle welds. In addition, circumferential cracking of CRDM nozzles in Alloy 600 base metal originating from the outside diameter (OD) of the nozzle has been identified. The cracking associated with safe end welds is important due to the potential for a large loss of coolant inventory, and the cracking of CRDM nozzle welds and circumferential cracking of CRDM nozzle base metal is important due to the potential for control rod ejection and LOCA.

The industry response in the U.S. to this cracking is being coordinated through the Materials Reliability Project (MRP) in a comprehensive, multifaceted effort. Although the industry program is addressing many of the issues raised by these cracking occurrences, confirmatory research is necessary to review the work conducted by industry groups. Several issues requiring additional consideration regarding the generic implications of these isolated events have been identified.

This paper will discuss significant events (i.e. V.C Summer, Ringhals, Oconee) of stress corrosion cracking (SCC), discuss the non-destructive examination (NDE) deficiencies in detecting SCC, discuss the NDE capabilities for control rod drive mechanisms (CRDMs), identify deficiencies in information available in this area such as crack initiation sites, crack growth rates; discuss United States Nuclear Regulatory Commission (USNRC) approach to address this issues, and the development of an international cooperative.

This page left blank intentionally

Regulatory Activities Related to Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

Allen L. Hiser, Jr.
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

The recent discoveries of cracked and leaking Alloy 600 vessel head penetration (VHP) nozzles, including control rod drive mechanism (CRDM) and thermocouple nozzles, at four pressurized water reactors (PWRs) have raised concerns about the structural integrity of VHP nozzles throughout the PWR industry. Nozzle cracking at Oconee Nuclear Station Unit 1 in November 2000 and Arkansas Nuclear One Unit 1 in February 2001 was limited to axial cracking, an occurrence deemed to be of limited safety concern in the NRC staff's generic safety evaluation on the cracking of VHP nozzles dated November 19, 1993. However, the discovery of circumferential cracking at Oconee Nuclear Station Unit 3 in February 2001 and Oconee Nuclear Station Unit 2 in April 2001 – particularly the large circumferential cracking identified in two CRDM nozzles at ONS3 – has raised concerns about the potential safety implications and prevalence of cracking in VHP nozzles in PWRs.

In response to the circumferential cracking identified at the Oconee units, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001. This bulletin requests information from licensees related to the structural integrity of the reactor pressure VHP nozzles for their facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, future plans to inspect VHP nozzles, and a description of how future inspection plans will ensure compliance with applicable regulatory requirements.

This paper summarizes the staff's review and assessment of licensee responses to NRC Bulletin 2001-01.

This page left blank intentionally

"Aging Evaluation of Cables in Japan"

Atsunori Yamaguchi
Tsurumi Research and Development Center
Japan Power Engineering and Inspection Corporation

30 years have already passed since the initial commercial nuclear power plants started operation in Japan. 51 plants are operated now ,and among them, Tsuruga power Plant Unit-1, Mihama Power Plant Unit-1 and Fukushima No.1 Nuclear Power Unit-1 have been operated over 30 years.

Under these circumstances, the Ministry of Economy, Trade and Industry (METI, former MITI) issued "Basic Policy on Aged Nuclear Power Plants" in April, 1996. According to the MITI report, electric utility companies performed the detailed technical assessments of nuclear power plant components when they reach 30 years of service life and established the detailed long-term maintenance schedules thereafter .

The final reports concerning the technical assessments conducted by the three electric utility companies on 3 nuclear powers plant mentioned were issued in February 1999. Aging of cables were evaluated in these reports because polymer materials used as insulating materials of cables were though to be aged by heat and radiation.

Safety-related cables installed within the containment vessels of nuclear power plants were classified according to the types (high voltage cable, low voltage cable, co-axial cable) and insulating materials. Technical Assessments were performed for the representative cables of each groups in accordance with "Recommendation methods on environmental qualification tests and tests for fire prevention of electric cables used in nuclear power plants " (The Japanese Electrotechnical committee technical report (II) No.139) based on IEEE Std. 323 & 343. Tests were performed under the conditions which enveloped the 60-year normal operating conditions. Examples of test conditions for high voltage cables were shown in the following table.

Table: Test conditions for technical assessments of cables

Test conditions	Tsuruga Unit-1	Mihama Unit-1	Fukushima No.1 Unit-1
Temperature	394K-7days	418K-32days	394K-7days
Total dose	500kGy	450kGy	500kGy

And for Cables which must fulfill the function during LOCA, tests were performed adding the LOCA conditions. Mandrel bend tests were done for the aged test samples and no break of insulation resistance occurred. These test results showed that cables could fulfill the necessary functions after 60- year operation.

By the way, the Recommendation methods mentioned above was issued in 1982. Since then, Many studies have been performed on the aging evaluation of cables and the following new knowledge have been obtained in Japan.

(1) Aging mechanism and accelerated aging technique by irradiation

Polymer materials for insulators are degraded by irradiation because of the radiation-induced oxidation. This radiation-induced oxidation mechanism was made clear, namely, the radiation-induced oxidation depended on the dose rate and the oxygen diffusion rate into the polymer materials. The relationship between the thickness of the oxidation layer and irradiation conditions is expressed in the following equation.

$$L=(2DS/\Phi)^{1/2}(P/I)^{1/2} \qquad (1)$$

L: Oxygen penetration thickness (cm), P: Oxygen pressure (MPa), I: Dose rate (Gy/s), D: Diffusion coefficient of oxygen into polymer material (cm^2/s), S: Solubility coefficient of oxygen into polymer material (mol/g/MPa), Φ: Specific oxygen consumption by radiation oxidation (mol/g/Gy)

This equation indicates that the thickness (L) increases with the decrease of dose rate and vice versa. So, the thickness (L) necessary to simulate the radiation aging can be obtained by adjusting the ratio of oxygen pressure (P) and dose rate (I).

(2) Multiplication of thermal aging and radiation aging

Radiation-induced oxidation was observed to progress by heating after irradiation, and the following relationship concerning the degree of aging was made clear.

thermal aging after radiation aging

>simultaneous aging by irradiation and heating

>radiation aging after thermal aging

(3) Accelerated thermal aging technique

It is usual that the thermal aging in the operational temperature range is estimated by using the activation energy obtained in the relatively high temperature range. But the experimental data showed that the act ivation energy in the lower temperature range was lower than that in the higher temperature range. It was observed that thermal oxidation occurred uniformly over the cross section of insulating material in the lower temperature range, though thermal oxidat ion occurred mainly near the outer surface region in the higher temperature range. This phenomenon seemed to be one of the reasons why the activation energy differed.

In near future, we will set appropriate accelerated procedures for the aging evaluation of cables used in the containment vessel of nuclear power plant, based on the new knowledge.

Enhanced Airworthiness Program for Airplane Systems

Massuod Sadeghi, Aging Systems Program Manager
Federal Aviation Administration

Safety concerns about aging wiring systems in airplanes were brought to the forefront of public and governmental attention by an accident involving a Boeing Model 747-131 airplane, operated as Trans World Airlines Flight 800, on July 17, 1996. That accident prompted the FAA to initiate investigations into fuel tank wiring, and to strengthen its focus on aging wiring in general. To add to existing knowledge about aging wiring, the aging structures program, already in place at the FAA, was expanded to include a systems component named the Aging Transport Non-Structural Systems Plan. The FAA has collected data in cooperation with industry, and other aviation authorities and developed actions based on this data. The FAA is in process of implementing these actions through the Enhanced Airworthiness Program for Airplane Systems (EAPAS). Age is not the sole cause of the wire degradation we call "aging." The probability that inadequate maintenance, contamination, improper repair, or mechanical damage has occurred to a particular wiring system increases over time. EAPAS is designed to enhance the existing airworthiness programs to mitigate the effect of these problems.

This page left blank intentionally

Methods to Integrate Aging Effects into Probabilistic Risk Assessments

Arthur Buslik
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

This paper will review some past work on the incorporation of hardware aging into probabilistic risk assessments (PRAs), and will describe ongoing work on the development and application of methods for incorporating aging into PRA. Ref. 1 (NUREG/CR-6157, 1994) reviews some of the earlier work related to the incorporation of hardware aging into PRA, and proposed an approach for including hardware aging in PRA which used reliability physics models to assess the probability of failure of hardware items. The use of reliability physics models helps mitigate the problem of sparseness of failure data relevant to the aging of hardware. Although some earlier work on the risk aspects of hardware aging considered active components and systems, the recent work has focused on long-lived passive systems, structures and components (SSCs), since the risk of hardware aging seems to be predominantly associated with these SSCs. Active systems are tested periodically and are overhauled or replaced at regular intervals. The approach suggested in Ref. 1 was investigated further in Ref. 2 (NUREG/CR-5632, August 2001), where a feasibility study was performed, for the particular case of flow accelerated corrosion in a main feedwater system of a PWR. The approach incorporated a careful treatment of epistemological and aleatory uncertainties. The approach uses the time dependent failure probabilities obtained from the reliability physics models and incorporates these failure probabilities into the fault and event tree models used in the PRA. The event tree and fault tree models are solved at various time points, to determine the core damage frequency as a function of time. Risk to the public could also be calculated, as could large early release frequencies.

The use of reliability physics models is resource intensive, and hence is best applied only to those items whose aging is risk significant. Various methods have been used to select the items whose aging should be modeled. The most promising way seems to be the application of importance measures. The importance of a particular item to the core damage frequency can be described by the conditional probability of core damage given the failure of the hardware item times the probability the hardware item will fail. Since we are interested in the effects of aging, this quantity could be estimated at, say, the end of life for the power plant. In some cases, there could be common cause failure of redundant components because of aging. In such cases it may be best to use a conditional probability of core damage given the set of components failed, times the probability the set of components fail. For the purposes of screening, conservative estimates can be used. In cases where the conditional probability of core damage given the failure of the component is appreciable, it is necessary to have a preliminary, conservative, estimate of the failure probability of the component, and how this depends on the component age. This could come from expert opinion, available data on component failures, and simplified reliability physics models. If the conditional probability of core damage given the failure of the component is sufficiently small, it is unnecessary to estimate the failure probability of the component.

The application of reliability physics models to aging reported in ref. 2 did not incorporate the results of inspection, but it is clear that, in general, this must be included for realistic estimates of the risk. If, say, a pipe is detected to be thinned excessively by flow-accelerated corrosion, then it will be replaced. The possibility of this replacement should be included in the PRA models. However, calculations which do not include the results of inspection are useful in identifying the components whose inspection would reduce the aging risk most.

Current work on the incorporation of aging into PRA models is focused on cable aging in harsh environments, in particular inside reactor containments. One is interested here in the likelihood of cable failures (or cable systems, including splices and terminations) after a loss of coolant accident (LOCA), and in the probability of core damage, given the cable failures. It is generally believed that the likelihood of the failure of instrumentation and control cables is greater than that of power cables, and current work is focused on this problem. There are some interesting features of this problem. First of all, there is the potential for common mode failure of cables associated with redundant systems. Secondly, whereas in the problem considered in reference 1, the effect of failure of a component could be modeled fairly straightforwardly, here the effect of failure of a cable may be the reporting of misleading or incorrect instrumentation reading to the control room operators, and it is necessary to estimate the human error probability under these circumstances. This is in itself a difficult problem. Thirdly, there is little information from which to directly estimate the failure probability of the cables in a LOCA environment. The information on cable failures during normal operation is not very relevant, since the environment is different. The accelerated aging tests performed for qualifying the cables cannot be used to estimate a failure probability, since only the reports of a successful test are required. There is some limited testing of cables which may be useful in estimated the failure probabilities. There do exist Arrhenius models and estimates of activation energy, and models for aging due to radiation dose, which can be used to correlate a cable effective age with the qualified lifetime of a cable. The time after the accident that the cable fails is also important; failures of instrumentation cables a few days after an accident may not have much risk significance, since the important human actions have already taken place and there is more time for recovery actions.

References

1. D. Sanzo et al., "Survey and Evaluation of Aging Risk Assessment Methods and Applications", NUREG/CR-6157, November 1994.

2. C.L. Smith et al., "Incorporating Aging Effects into Probabilistic Risk Assessment -- A Feasibility Study Utilizing Reliability Physics Models", NUREG/CR-5632, August 2001.

AGE-RELATED DEGRADATION OF STRUCTURES AND PASSIVE COMPONENTS AT NUCLEAR POWER PLANTS

J.I. Braverman, C.A. Miller, C.H. Hofmayer - Brookhaven National Laboratory, Upton NY
B.R. Ellingwood - Georgia Institute of Technology, Atlanta GA
D.J. Naus - Oak Ridge National Laboratory, Oak Ridge TN
T.Y. Chang - US Nuclear Regulatory Commission

This summary describes the multi-year research program to assess age-related degradation of structures and passive components important to the safe operation of nuclear power plants (NPPs). The purpose of the research effort is to develop the technical basis for the validation and improvement of analytical methods and acceptance criteria which can be used to make risk-informed decisions and to address technical issues related to degradation of structures and passive components. The approach adopted for this research program consists of three Phases. In Phase I, specific degradation occurrences at plants were collected and evaluated, existing technical information on aging was reviewed, and a scoping study was performed to identify which structures and components should be studied in the subsequent phases of the research program. Based on the results of the Phase I effort, selected structures and passive components are evaluated in Phase II to assess the effects of age-related degradation using existing and enhanced analytical methods. Phase III utilizes the results of the analyses to develop recommendations to the NRC staff for making risk-informed decisions related to degradation of structures and passive components.

The Phase I assessment of age-related degradation of structures and passive components has been completed and is reported in NUREG/CR-6679. This assessment consisted of three activities. In the first activity, instances of age-related degradation were collected and evaluated. The data were collected from LERs, NRC generic communications, NUREG reports, and industry reports. A computerized database was developed to summarize important parameters which describe the applicable cases of degradation. Trending analyses were performed to identify which structures and passive components are most susceptible to age-related degradation, whether degradation occurrences are increasing, and other important observations. In the second activity, additional information such as NRC requirements/guidance, NRC programs, industry programs, degradation information from other countries, and other reports/papers on aging degradation were evaluated to identify the significant aging issues for those structures and passive components which could have the greatest impact on plant risk. In the third activity, collection of degradation occurrences, trending analyses, available technical information, and risk significance of aging effects were utilized in a scoping study to identify those structures and passive components that should be studied in Phase II of this program. The structures and passive components were prioritized to determine which items warrant further detailed evaluation in Phase II. The scoping study concluded that the structures and passive components that should be given highest priority for further evaluation are masonry walls, flat bottom tanks, anchorages, concrete structures (other than containment), and buried piping. This list is based on the knowledge that other NRC and industry programs already exist to address items such as containment, steam generators, RPV internals, and piping.

Phases II and III of the research program is currently in progress. The first of five structures and passive components selected was reinforced concrete members. The results of the evaluation for degraded reinforced concrete members are reported in NUREG/CR-6715. The objective of the research was to develop analytical methods and acceptance limits for degraded reinforced concrete members. Results from risk evaluation programs conducted by the NRC, such as the Individual Plant Examination of External Events (IPEEE) program, show that external events can be significant contributors to core damage frequency (CDF). In some cases, structures and passive components have been found to be significant risk contributors when subjected to external events such as earthquakes. Therefore, the research program focused on developing fragility models for evaluation of degraded reinforced concrete members subjected to earthquake forces. These analytical methods were used to develop probability-based degradation acceptance limits based on the impact of degradation on overall plant risk. The objectives of the program were achieved by performing four major activities: evaluation of degradation mechanisms and condition assessment methods for reinforced concrete, structural evaluation of degraded concrete members, fragility and risk evaluation of degraded concrete, and development of probability-based degradation acceptance limits.

The research effort developed fragility modeling procedures for undegraded and degraded reinforced concrete flexural and shear wall members subjected to earthquake ground motions. The research also provided the technical basis for developing probability-based degradation acceptance limits. These results provide a basis for evaluating degraded reinforced concrete structures in nuclear plants for continued service and for developing guidelines for in-service inspection and repair. The probability-based degradation acceptance limits that have been developed can be used as a tool for making risk-informed decisions regarding degradation of reinforced concrete members.

The second structure or passive component selected for evaluation in Phases II and III is buried piping. The approach described above for reinforced concrete members is currently being applied to the evaluation of buried piping and will be utilized for the other three remaining structures/components. This will involve developing methodologies for performing structural analyses of undegraded and degraded structures/components, conducting fragility and risk evaluations, relating the level of degradations to observable manifestations of degradation, and recommending probability-based degradation acceptance limits based on these relationships. The acceptance limits developed in this research can be used during plant inspections to evaluate whether age-related degradation of the structure/component potentially has a significant effect on plant risk. The probability-based degradation acceptance limits provide a tool for making risk-informed decisions regarding the suitability of a degraded structure or passive component to continue service with or without repair.

Study of High Burnup Fuel Behavior under LOCA Conditions at JAERI: Hydrogen Effects on the Failure-bearing Capability of Cladding Tubes

Fumihisa Nagase and Hiroshi Uetsuka
Japan Atomic Energy Research Institute

The Japanese LOCA criteria on fuel safety, 15% cladding oxidation (ECR, equivalent cladding reacted) and 1200 °C peak cladding temperature, were established in 1975 and were based on the concept of zero ductility of cladding as in the U.S. After their establishment, the Japan Atomic Energy Research Institute (JAERI) found that inner surface oxidation after rod-burst is accompanied by significant hydrogen absorption [1]. Ring compression tests were performed on the cladding specimens that experienced rod-burst and double-sided oxidation to examine the embrittlement of the cladding due to oxidation and hydrogen absorption. The ductility of ring specimens fell down to the zero ductility range when the cladding was oxidized to several percent ECR, indicating that significant hydrogen absorption enhances cladding embrittlement [2]. Accordingly, JAERI conducted "integral thermal shock tests" to evaluate the failure-bearing capability of oxidized cladding under appropriately simulated LOCA conditions [2]. In the test, a short test rod was heated up, burst, oxidized in steam and quenched by flooding water, and it was completely restrained during quenching in order to conservatively simulate the possible occurrence of tensile loads on a fuel rod due to axial shrinkage of the cladding. Obtained results confirmed that the criterion of 15% ECR still had safety margin, and the LOCA criteria were revised in 1981 referring the results of the integral thermal shock tests. Therefore, the current Japanese LOCA criteria on fuel safety are not based on the concept of zero ductility of cladding, but on the failure threshold value determined in the integral thermal shock tests under restrained conditions. However, the current LOCA criteria are generally based on a database from tests with non-irradiated cladding materials.

With a view to obtaining basic data to evaluate high burnup fuel rod behavior under loss-of-coolant accident (LOCA) conditions, a systematic research program is being conducted at the Japan Atomic Energy Research Institute (JAERI). Since the integral thermal shock tests are essential to evaluating the safety of the high burnup fuel as described above, the program consists of integral thermal shock tests and other separate tests including Zircaloy-steam oxidation tests, mechanical property tests of cladding, and tube burst tests. Several types of Zircaloy cladding samples are used for these tests to provide a wide range of basic data available for regulatory judgment. They are (a) as-received cladding tubes, (b) simulated high burnup fuel cladding that is artificially pre-oxidized, pre-hydrided and/or neutron irradiated, and (c) high burnup PWR fuel cladding [4].

Key points for evaluating the failure-bearing capability of the high burnup fuel cladding are:

- Degradation of cladding ductility due to reduction of cladding wall thickness by waterside corrosion and hydrogen absorption
- Axial constraint condition

Considering these points, the following integral thermal shock tests are being performed with non-irradiated cladding tubes in the present study. The cladding tubes were mechanically thinned and pre-hydrided to simulate corrosion and hydrogen absorption of high burnup fuel cladding. About 10% reduction of initial cladding thickness and a hydrogen concentration of 100 to 1000 ppm were adopted.

A test rod consisting of the cladding tube and Alumina pellets are isothermally oxidized at 1000 through 1250 °C for 30 to 5500 sec after rod-burst. Both ends of the test rod were fixed just before quenching to simulate the possible tensile loads on the cladding tube. Since the test condition of the above-mentioned thermal shock test [3] was probably too conservative in terms of constraint condition, the tensile load was controlled and limited to three different levels of about 390, 540, and 735N (40, 55 and 75kgf) in addition to the fully restrained condition to realize intermediate constraint conditions during quenching.

As a result, variations of the failure threshold value were evaluated as functions of hydrogen concentration and restrained condition during quenching. The influence of pre-hydriding was obviously seen on the failure threshold value under severer loading conditions. The threshold value of pre-hydrided cladding tubes was as low as 10% ECR under the fully restrained condition, which is the most conservative loading condition. The failure threshold generally increased with the decrease in controlled tensile load, and it was estimated to be higher than 20% ECR when the tensile load was controlled below 540N (55kgf) for the hydrogen concentration range that was examined.

References

1. H. Uetsuka et al., J. Nucl.Sci.& Tech. Vol.20, No.9, pp.705, 1981.
2. H. Uetsuka et al., report JAERI-M 9445, April, 1981 (Text in Japanese).
3. H. Uetsuka et al., J. Nucl.Sci.& Tech., Vol.20, No.11, pp 941, 1983.
4. F. Nagase et al., Proceeding of ANS topical meeting on LWR fuel performance, April 10-13, 2000, Park City.

Overview of Test Results on Mechanical Properties of Unirradiated and Irradiated Zr–1%Nb E110 Alloy Cladding

L. Yegorova, E. Kaplar, K. Lioutov
Nuclear Safety Institute of the Russian Research Center "Kurchatov Institute"
Moscow, Russian Federation

V. Smirnov, A. Goryachev
State Research Center "Research Institute of Atomic Reactors"
Dimitrovgrad, Russian Federation

Studies of commercial Zr–1%Nb cladding made of the E110 alloy were performed to reveal differences in the mechanical behavior of unirradiated and irradiated cladding under accident conditions. The research program included several types of mechanical tests:

- Uniaxial tensile tests in the transverse direction (293 - 1200 K);
- Uniaxial tensile tests in the axial direction (293 - 1200 K);
- Low temperature biaxial burst tests with and without axial loading (293 - 1200 K);
- High temperature burst tests (293 - 1400 K).

Cladding samples for these tests were manufactured from the as-received Zr–1%Nb tubes and irradiated Zr–1%Nb cladding from VVER high burnup fuel rods (~ 48 MWd/kgU). The tests were carried out so that the mechanical properties of irradiated and unirradiated cladding specimens were obtained versus the temperature, strain rate, direction of loading (transverse and axial), type of loading (uniaxial and biaxial), level of biaxiality (ratio of 1 and 2).
The final data base with the test results contains the following mechanical properties:

- The yield stress, ultimate strength, uniform elongation, total elongation (for uniaxial tests);
- The burst pressure, burst strain, burst stress (for biaxial tests).

In addition, the test results were used to develop a plastic deformation correlation and to estimate the anisotropy coefficients for irradiated and unirradiated Zr–1%Nb cladding.

Comparative analysis of the data show that Zr–1%Nb cladding exhibits some general tendencies that are unaffected by the type of loading:

- Irradiation of Zr–1%Nb cladding led to an increase in cladding strength and a decrease in cladding ductility in the low temperature range;
- The difference between mechanical properties of unirradiated and irradiated cladding disappeared completely at temperatures higher than 860 K;
- The peak value of the cladding strain (up to 100 %) was observed at the temperature of 1000 K for irradiated and unirradiated cladding.

The consideration in more detail of the mechanical behavior of tested cladding versus such factors as the loading direction, biaxiality ratio showed the following:

- The anisotropy effect was insignificant for irradiated cladding within the temperature range 293–1400 K and for unirradiated cladding at temperatures higher than 800 K;
- The strength parameters for unirradiated and irradiated cladding were practically independent of the biaxiality ratio (1 or 2);
- A high sensitivity of the circumferential elongation to the biaxiality ratio was substantiated for unirradiated cladding within the low temperature range 293–723 K;
- The circumferential elongation of irradiated cladding within the temperature range 293–1400 K was independent of the biaxiality ratio.

HIGH TEMPERATURE OXIDATION OF IRRADIATED LIMERICK BWR CLADDING

Y. Yan, R. Strain, T. Bray, and M.C. Billone

Argonne National Laboratory
Argonne, Illinois, USA

SUMMARY

The High Burnup Cladding Performance program is being conducted at Argonne National Laboratory to provide data that support modeling efforts and that assess licensing criteria for Loss of Coolant Accident (LOCA) and Reactivity-Induced Accident (RIA) events involving high burnup fuel rods. The program is sponsored by the USNRC Office of Nuclear Regulatory Research. The Electric Power Research Institute (EPRI) also plays a major role in the program planning and conduct and in supplying the high burnup BWR and PWR fuel rods to ANL: seven Limerick BWR fuel rods (≈57 GWd/MTU) and seven H.B. Robinson PWR fuel rods (≈67 GWd/MTU). The major tasks of the program include: fuel and cladding characterization, cladding high temperature steam oxidation kinetics study, LOCA-relevant testing, and mechanical properties (uniaxial tensile, plane strain, biaxial, and bending). The focus of this paper is the results of the oxidation kinetics study and their impact on LOCA Integral Test planning and interpretation of results.

The current LOCA licensing criteria (10 CFR50.46) limit peak cladding temperature to 2200°F (1204°C) and maximum Equivalent Cladding Reacted (ECR) to 17% during high temperature steam oxidation to ensure adequate ductility during the Emergency Core Cooling System quench and during possible post-LOCA seismic events. In addition, NRC Information Notice 98-29 specifies that the ECR should be based on the total oxidation, including oxide layers formed during normal reactor operation. For PWR cladding, high burnup operation may induce coolant-side oxidation thicknesses of ≈100 μm, corresponding to 10-14% ECR. This would leave very little margin for the LOCA transient oxidation. The primary high burnup phenomena that may affect cladding response during ballooning and burst, steam oxidation, quench and post-quench are: loss of base metal thickness, hydrogen pickup (500-700 wppm at ≈100 μm oxide thickness) and formation of an inner-surface oxide layer, all during normal operation; the effective thickness and chemistry (i.e., H and O content) of the prior β-phase layer following steam oxidation and quench; and decreased fuel permeability and the tightness of the fuel-cladding bond. The LOCA Integral Tests will be conducted with high burnup fueled cladding segments in order to include all the phenomena highlighted. However, it is essential that oxidation studies be performed with cladding samples from the high burnup rods to plan the LOCA Integral Test experimental times that will test the adequacy of the current criteria and will determine the failure threshold for fragmentation during quench and/or nil-ductility following quench.

The test plan for oxidation studies of high burnup BWR and PWR cladding specifies ranges of temperature (900-1300°C) and test times (0-300 minutes). There are two main purposes for these tests: to provide adequate oxidation data at 1204°C (0-20 minutes) to achieve ECR values ≤ 30% for LOCA-criteria test planning; and to develop fundamental data for modeling codes on the effects of high burnup operation on high temperature steam oxidation kinetics. Of particular interest in these studies is the influence of the in-reactor-formed oxide layer, and associated hydrogen pickup, on the oxidation kinetics and phase boundary evolution during steam

oxidation. In order to determine the effects of these parameters on oxidation kinetics, unirradiated archival cladding samples are tested in the same apparatus used to test the high burnup samples. One-sided oxidation tests are conducted to determine the oxidation kinetics. The experimental results for the irradiated Limerick BWR cladding are presented, along with the baseline data for the archival Zircaloy-2 tubing and for Zircaloy-4 tubing.

At the 28th WRSM, weight gain results were presented for unirradiated Limerick archival Zircaloy-2 tested in steam at 1204°C for 5, 10, 20 and 40 minutes. Three independent methods were used to determine weight gain: change in total sample (25-mm-long) weight normalized to the oxidation surface area, oxygen content determined from metallography and model calculations, and change in oxygen content determined from direct LECO measurements. Relative to the predictions of a best estimate model (Cathcart-Pawel), the total sample weight gain data were high due to end effects and interior regions of nonuniform oxide growth; the weight gains deduced from the metallography were in excellent agreement with model predictions and the weight gains determined from the direct measurement of oxygen concentration were low due to oxide/alpha material loss during sample preparation. Also, the oxide layer thickness data from the metallography were in excellent agreement with the Cathcart-Pawel predictions.

Following the 28th WRSM, the companion tests on irradiated Limerick Zircaloy-2 cladding were analyzed. Qualitatively, the results of the three independent methods for determining weight gain were similar to those results for the unirradiated cladding. However, the oxide layer thicknesses measured – and hence the weight gains deduced from the metallography – were about 40% larger than those predicted by the Cathcart-Pawel model. As a result of these experiences, the oxidation apparatus was redesigned to provide better control of the test chamber environment, to provide higher steam flow rates, and to provide more uniform steam flow rates and spatial distribution of the steam flow.

With the new oxidation apparatus, extensive out-of-cell thermal and oxidation-kinetics benchmark tests were conducted using unirradiated Zircaloy-2 and Zircaloy-4 samples. Ten additional data points have been generated for times ranging from 5 to 20 minutes and temperatures ranging from 1000-1204°C. The new test results indicate uniform oxide/alpha/beta layer thickness over about 90% of the length of the samples. The total sample weight gain data are ≈15% higher than the model predictions due to limited diffusion/double-sided oxidation at the sample ends. Improvements in sample preparation have resulted in the weight gains from the direct measurements of oxygen content to be ≈14% lower than model predictions – a significant improvement over previous results. As before, the oxide layer thickness data and weight gains deduced from metallography are in excellent agreement with Cathcart-Pawel model predictions.

Additional tests are being performed on the irradiated Limerick cladding at 1204°C (5-20 minutes) and at 900, 1000, 1100, and 1300°C. Data from these tests will be compared to the Cathcart-Pawel model predictions to determine if the oxidation kinetics for the irradiated cladding are faster, slower or about the same as for the unirradiated archival cladding.

Status of the CABRI International Programme
and Preparation for the CIP0 Test Series

J. Papin, C.Marquié, F.Lemoine, M.Faury, J.C. Melis

Institut de Protection et de Sûreté Nucléaire
Cadarache Research Center
Saint Paul lez Durance, France

The CABRI International Programme (CIP) is now being launched by the "Institut de Protection et de Sûreté Nucléaire (IPSN)" with the collaboration of Electricité de France (EDF) in the frame of a broad international cooperation under the auspices of the OECD. It has been initiated in the context of the world-wide evolution of the fuel management strategy with the further increase of the UO2 fuel discharge burnup and the introduction of the MOX fuel which created the need for qualification of the advanced fuels under reactivity initiated accidents (RIAs) as resulting from control rod ejection.

The main objectives of the CIP are to provide, under typical PWR conditions, bases for the assessment of new safety criteria and evaluation of safety margins relative to the use of advanced alloys (ZIRLO, M5, Duplex...), high burnup fuels and future fuel concepts including MOX. The basic test matrix is composed of twelve tests included into different series with the following general objectives :

CIP0 : two reference tests in the sodium loop with high burnup fuel and advanced cladding material (ZIRLO and M5)
CIPQ : Qualification test of the Water Loop
CIP1 : two reference tests in the water loop , similar to CIP0 tests
CIP2 : very high burnup fuel (80-100 GWd/t, one test with Duplex rod)
CIP3 : improvement of physical understanding of RIA
CIP4 : tests with MOX fuel
CIP5 : complementary tests (open)

Since last year, a more precise definition of the tests matrix has been elaborated through the work of the Technical Advisory Group (TAG) of the CABRI program.

A special focus has been given to the preparation of the two tests of the CIP0 series to be performed in the CABRI sodium loop in 2002 with the specific objectives of giving a first answer on the behaviour of UO2 advanced fuel rods at higher burnup than the current values and providing a link between sodium loop and water loop tests (CIP1). Representative power pulse width together with an energy injection of about 100 cal/g are anticipated : precise test conditions will be based on pre-calculations including parametric studies on pulse width and evaluation of rod failure risk.

The CIP0-1 experiment is being prepared using the span 5 of an UO2 ENUSA rod irradiated in Vandellos reactor up to 68 GWd/tU (rod average) with ZIRLO cladding (average measured corrosion thickness of 80µm at span 5, without spalling). The rod will be shipped to France in November 2001 after reconditioning and characterization (in Studsvik, Sweden) and loaded in the test device to be tested in CABRI in April 2002.

The CIP0-2 test is foreseen using the span 5 of an UO2 EDF rod irradiated up to 69 GWd/tU (rod average) in the Gravelines 5 reactor, with a M5 cladding (expected corrosion thickness about 30µm). Rod characterization and reconditioning are scheduled during first semester 2002 for a test planned in autumn 2002.

The CIP1 series will be performed in the water loop using twin rods of the CIP0 ones. The need of a qualification test for the water loop has been underlined and led to propose the CIPQ test as the first test to be performed in water conditions. Its specific objectives are to validate the facility operation throughout the whole transient including possible rod failure and check the instrumentation response. The use of a UO2 rod with Zr-4 cladding and moderate burnup (50 GWd/tU) is envisaged and prospective studies are underway at IPSN in connection with the evaluation of the representativity of the test channel as compared to reactor case (influence of annular/reactor geometry, spacer system with thermocouples). In case of success, such test addressing the RIA phenomenology, will be part of the CIP3 series.

The detailed definition of the other tests series is in an early stage. However, beyond the impact of fuel burnup, fuel and cladding types, objectives for various fields of investigation have been identified such as effects of initial power level, fuel duty under operation, power pulse width, understanding of failure mechanism and study of consequences of rod failure. In addition, several fuel rod candidates have been proposed by the partners and choice will be done on the basis of detailed objectives, common interest and availability.

As a general concern for the understanding and interpretation of the results, a standard pre and post test examinations program has been elaborated and is under finalization; the importance of the initial characterization of father rods (to be provided with the test rods) has been underlined.

In parallel to the test matrix definition, a minimum experimental program for mechanical characterization of the advanced cladding materials has been proposed. Its objectives are to provide clad mechanical constitutive laws, data for failure criteria and evaluation of the material anisotropy. Such a program is foreseen for each new cladding material experimented in the CABRI project and for a same material at different burnup levels if tests with significantly different burnup levels are realized. It is already defined for the ZIRLO material in relation with the CIP0-1 and CIP1-1 tests.

On the other side, important research and development work is launched in the field of instrumentation in order to implement high quality measurement needed for reliable quantification of the phenomena. An associated qualification program is elaborated and concerns sensors in pile capability with fast transient response under high temperature and pressure conditions.

The CABRI facility is now shut down for a first phase of renewal; the work needed for additional renewal will be fixed in end 2001. The water loop detailed studies have been completed and the implementation is scheduled in mid 2003-2004 with first acceptance tests in first trimester 2005. Such schedule has to be confirmed according to the expertise for renewal work and to safety authorizations based on a decree expected in mid 2003 and on an operating license in early 2005.

FRAPTRAN Fuel Rod Code and its Coupled Transient Analysis with the GENFLO Thermal Hydraulic Code

Keijo Valtonen[1], Anitta Hämäläinen[2], Mitchel E .Cunningham[3]

[1]Radiation and Nuclear Safety Authority, STUK, P.O. Box 14, FIN-00881 Helsinki, Finland
[2] VTT ENERGY, P.O. Box 1604, FIN-02044 VTT, Finland
[3]Pacific Northwest National Laboratory, P.O. Box 999, Richland, WA

Plans to increase the burnup of nuclear fuel, to utilize new fuel designs, and in some countries to include additional transients such as the anticipated transient without scram (ATWS) in safety evaluations, require that new or updated models be used in safety analyses. Addressing these issues requires improving the fuel models in reactor dynamics codes or incorporating more advanced thermal hydraulic models in fuel behaviour codes. An example of the latter approach involves coupling the Finnish thermal hydraulic model GENFLO (GENeral FLOw) with the new US Nuclear Regulatory Commission (NRC) FRAPTRAN code. This approach and two examples of results are presented in this paper.

FRAPTRAN is being developed and maintained for the NRC to calculate fuel behavior during power and cooling transients at burnup levels up to at least 62 GWd/MTU (Cunningham et al. 2001a). FRAPTRAN has been assessed (Cunningham et al. 2001b) using a data base that emphasized experiments investigating the effects of burnup on fuel rod behavior during reactivity-initiated accidents (RIAs) and loss-of-coolant-accidents (LOCAs). Assessment results have provided assurance that the basic models (temperature, gap conductance, gas pressure, and thermal expansion) are working correctly. For the LOCA cases, rod internal gas pressure was well calculated and predicted time to cladding failure was generally close to that observed in the LOCA tests. For the RIA assessment cases, FRAPTRAN predicted fuel axial thermal elongation well, thus indicating a good prediction of fuel temperatures. However, FRAPTRAN consistently underpredicted cladding permanent hoop strain relative to cladding permanent axial strain. This is indicative of radial fuel-cladding mechanical interaction occurring that is not modeled by the code.

In addition to the code assessment, an independent peer review of FRAPTRAN has been conducted. Principal conclusions include: the FRAPTRAN fuel and cladding models are reasonable, FRAPTRAN is able to predict the trends in the experimental data, and the assessment data base adequately covered the intended applications for the code. FRAPTRAN is being released though the FRAPCON-3 and FRAPTRAN users group managed by Pacific Northwest National Laboratory.

The hydraulic model in FRAPTRAN describes only homogeneous, slowly changing thermal hydraulic conditions and for many transients using a thermal hydraulic code is necessary to calculate coolant boundary conditions. Especially during the ATWS in BWR plants, the hot channel and the whole core may experience rapid transitions between the wetted and dry states. Because of this, a dynamic exchange of detailed local data between the fuel performance and thermal hydraulic models is needed.

The thermal-hydraulic model GENFLO has been developed by VTT. GENFLO is a fast running, five-equation model, where the wetted wall heat transfer, dryout, post-dryout heat transfer and quenching models are included. In coupling with FRAPTRAN, GENFLO calculates the thermal hydraulic behaviour of the fuel subchannel under transient conditions.

The system behaviour and boundary conditions needed for a detailed core simulation and study of the fuel rod behaviour in FRAPTRAN-GENFLO analyses may be calculated with various system codes such as RELAP5 or others. The boundary conditions for the hydraulic model from the system code are the mass flow and enthalpy at the channel inlet, the pressure at the top of the channel, and the total power and power profile of the fuel rod. The coolant mass, momentum and energy conservation equations are solved in GENFLO, including the calculation of axial distributions of fluid temperature and void fraction. As a result, the fluid temperatures and heat transfer coefficients for each axial level at each time step are supplied for FRAPTRAN. FRAPTRAN calculates the temperatures of the fuel rod, and the fuel and cladding deformation, including ballooning. At this stage GENFLO and FRAPTRAN use their own models, i.e., for fuel and cladding temperatures including cladding oxidation and hydrogen generation, though FRAPTRAN supplies the local gas gap heat transfer coefficient for GENFLO.

The analysis performed using FRAPTRAN-GENFLO was fuel behaviour during a hypothetical large break LOCA at the Loviisa VVER-440 power plant. The hot channel analysis was performed using APROS to supply the boundary conditions needed for the FRAPTRAN-GENFLO analysis. As a comparison to the effects of using GENFLO thermal hydraulics, the same case was calculated by FRAPTRAN using the simple built-in coolant model. No cladding ballooning was predicted with the coupled codes, but the separate FRAPTRAN calculation predicted ballooning, cladding deformation and rod failure.

A second analysis was for an ATWS at a BWR plant. The basis for the analysis was a real oscillation incident in the Finnish Olkiluoto 1 BWR during reactor startup on February 22, 1987. The incident was safely terminated by normal operation of the reactor safety systems. The incident was successfully simulated with the Finnish three-dimensional neutronics code TRAB-3D, showing good agreement with the real measurements. The case was then recalculated as an ATWS. The increasing oscillation phase of this calculation was chosen for study using FRAPTRAN-GENFLO. The oscillating boundary conditions were artificially continued in time and enlarged, which leads to temporary core inlet flow reversals. A changing axial power profile was also created according to the TRAB-3D calculations. First results of this assumed transient show that after the flow reversals the hot channels do experience transitions from the wetted state to local dryout and back.

The results of the example cases show that GENFLO can be used as a new thermal-hydraulic model for FRAPTRAN. Further, the ATWS instability case assumed for the second analysis needs more study. From these two cases, it may be concluded that FRAPTRAN-GENFLO may be effectively used for single fuel rod and subchannel analyses.

REFERENCES

Cunningham, M.E., et al. 2001a. *FRAPTRAN: A Computer Code for the Transient Analysis of Oxide Fuel Rods.* NUREG/CR-6739, Vol. 1, PNNL-13576. Pacific Northwest National Laboratory, Richland, Washington.

Cunningham, M.E., et al. 2001b. *FRAPTRAN: Integral Assessment.* NUREG/CR-6739, Vol. 2, PNNL-13576. Pacific Northwest National Laboratory, Richland, Washington.

Pulse Width During a PWR Rod Ejection Accident

D.J. Diamond, B.P. Bromley, and A. Aronson
Energy Sciences & Technology Dept.
Brookhaven National Laboratory
Upton, NY

This study is part of a U.S. Nuclear Regulatory Commission program to understand the consequences of reactivity accidents, especially for fuel with high burnups, and to define new acceptance criteria for these events. The work at Brookhaven National Laboratory (BNL) has focused on analytical studies of the neutronics and thermal-hydraulics during a rod ejection accident (REA) in a pressurized water reactor (PWR). Of particular interest have been parametric studies to determine what influences the increase in local fuel enthalpy (pellet radial average), the uncertainty in fuel enthalpy, and the characteristics of the power trace during the REA. The parameters of interest have been the ejected rod worth, delayed neutron fraction, fuel specific heat, initial power level, time during the fuel cycle, and location of the ejected rod. The calculated characteristics of the power pulse assure that corresponding experimental programs are designed with conditions that come as close as possible to those expected during the REA. Power pulses can be shaped by varying the pulse height and width (within limits) in the experimental programs in France, at the Cabri reactor, and in Japan, at the NSRR reactor. It is specifically the power pulse width which is of interest in the current study.

Calculations were carried out at BNL to determine the expected full width at half maximum (FWHM) using the PARCS code. The neutron kinetics in PARCS is treated in three dimensions and includes thermal-hydraulic feedback from the fuel and moderator. The reactor model was for the TMI-1 unit at end-of-cycle. The calculations assumed that the central control rod was ejected from hot zero power conditions with different reactivity worths obtained by changes to the cross section data for the central fuel assembly. The calculated power pulse was terminated initially by fuel temperature feedback and then power decays due primarily to reactor trip.

The results showed that pulse width varied inversely with the maximum increase in local fuel enthalpy. These results were independent of delayed neutron fraction and are expected to have a small dependence on reactor design. Hence, if one is interested in studying large fuel enthalpy increases, then one needs to consider narrow power pulses, e.g., 10-20 ms for FWHM to study enthalpy increase in the range 100-40 cal/g. These pulse widths would correspond to low probability REAs at close to what might be suggested as a regulatory limit. Conversely, if one were interested in a more probable REA that yielded a smaller enthalpy rise, then one needs to consider broader pulse widths, e.g., 20-60 ms to study fuel enthalpy rise in the range 40-15 cal/g.

This page left blank intentionally

The Industry-proposed RIA Criteria for Burnup Extension

Nicolas Waeckel, Electricite de France
Robert Montgomery, ANATECH Corp.
Rosa Yang, EPRI

The US nuclear industry through Working Group 2 of the Robust Fuel Program has developed for NRC review a set of revised regulatory acceptance criteria for use in the safety analysis of the hot-zero power (HZP) and hot-full power (HFP) Reactivity Initiated Accidents (RIA) in Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). For PWRs, the postulated control rod ejection accident (REA) is the primary RIA event considered in the industry evaluation. For BWRs, the postulated control rod drop accident (CRDA) is the primary RIA event considered in the industry evaluation. The revised RIA regulatory acceptance criteria are proposed as part of the on-going industry effort to extend fuel rod average burnup levels to 75 GWd/MTU for PWRs and 70 GWD/MTU for BWRs.

The approach to develop the revised criteria used an evaluation methodology that combined both experimental data and analytical calculations to understand the influence of burnup on transient fuel rod behavior during RIA events. Experimental data from RIA-simulation tests on UO_2 test specimens with Zircaloy cladding irradiated to 64 GWd/MTU were used in the evaluation. These technical bases were then translated to PWR REA applications using a state-of-the-art fuel rod behavior analysis code as a means to establish the revised regulatory criteria. The advantage of using a combined analytical and experimental data approach to derive the revised regulatory criteria is that the methodology can be applied to other cladding designs and fuel rod analysis tools to determine application specific criteria, if required.

Two separate criteria have been developed to 1) ensure long-term cooling of the reactor core after the accident and 2) account for radiological release to the environment following cladding failure. To assure core coolability and to preclude damage to the reactor pressure vessel, a core coolability limit is established based on the maximum radial average peak fuel enthalpy that precludes incipient UO_2 pellet melting during power deposition. Second, a threshold on the radial average peak fuel enthalpy is defined that represents the occurrence of fuel rod failure for use in off-site dose calculations. The fuel rod failure threshold below a rod average burnup of 35 GWd/MTU is established to preclude cladding failure by high temperature processes. Beyond a rod average burnup of 35 GWd/MTU, the fuel rod failure threshold is based on cladding failure by PCMI. Both the core coolability limit and the fuel rod failure threshold are defined as a function of fuel rod average burnup.

The proposed RIA regulatory acceptance criteria are applicable to Zircaloy-clad UO_2 or UO_2-Gd_2O_3 fuel rods operated up to a target lead rod average burnup of 75 GWd/MTU that experiences a HZP or HFP RIA with power pulse widths greater than 20 milliseconds. In the application of these criteria, the maximum cladding outer surface zirconium oxide layer thickness should not exceed 100 microns and there should be no oxide spallation that significantly impacts the cladding mechanical properties. The proposed RIA fuel rod failure threshold is applicable to advanced cladding designs provided the cladding material exhibits superior or equivalent ductility as Zircaloy cladding with the same outer surface oxide layer thickness and without oxide spallation. Future RIA-simulation experiments with UO_2 fuel rods will provide confirmatory data at high burnup and provide additional characterization of advanced cladding material.

Needs For Experimental Programs on LOCA Issues Using High Burn-up and MOX Fuels

Alain Mailliat, Michel Schwarz

Institut de Protection et de Sûreté Nucléaire
Cadarache Research Center
Saint Paul lez Durance, France

Safety studies performed in IPSN and elsewhere pointed out that high burn up might induce significant effects, especially those related with fuel relocation during LOCA situations. Uncertainties exist regarding how much the existing safety margins associated with peak clad temperature, clad oxidation, core coolability, clad residual ductility can be reduced by new fuels like the MOX one, burn up increases, the arrival of various alloys for fuel rod cladding. A better knowledge of specific phenomena associated to fuel effects is required in order to estimate the new margins and to resolve the pending uncertainties related to the LOCA criteria. Therefore, in addition to the programs currently planned in the Halden reactor, IPSN is preparing the so-called "APRP-Irradié" (High Burn up fuel LOCA) program. One of the important aspects of this program is In-Pile experiments involving bundle geometries in the PHEBUS facility located at Cadarache, France.

The IPSN presentation first addresses the uncertainties and pending questions related to the LOCA issues in presence of high burn up and MOX fuel:

1) Clad Deformation
- Influence of hydrogen pick-up and other irradiation effects on ballooning and burst behavior?

2) Fuel relocation
- Instant of fuel movement at high burn-up, with possible delay due to fuel-clad bonding?
- Filling ratio of clad balloon at high burn-up, with fragmentation of UO2 rim or MOX clusters?
- Effect on peak clad temperature and final oxidation ratio of the relative increase of the local heat load and of the decrease of the local fuel-clad gap resulting both from fuel accumulation in the balloon? This last question is particularly important for end-of-life MOX fuel where power generation is not reduced, unlike for UO2 fuel.

3) Flow blockage
- What is the maximum flow blockage ratio that remains coolable with irradiated rod bundle experiencing fuel relocation?

4) Embrittlement of high BU fuel rod
- Potential for mechanical load on cladding due to fuel swelling under fission gas expansion due to overheating in central fragments?

- Potential for PCMI constraints upon quench, when clad balloon is closely filled with fine fuel fragments?

In a second part of the paper, IPSN explains why, in addition and in a complementary way to the existing research programs, in-pile tests with bundle geometry are required for solving the previously mentioned points.

- In pile tests provide the unique way to maintain the correct heat generation, in the fuel fragments whatever the relocations induced by the ballooning and/or the burst of the rod.

- This heat generation correctness is one of the essential conditions for having realistic estimates of the relocation consequences in terms of equivalent clad reacted, peak clad temperature and hydrogen uptakes inside the balloon

- In-pile tests including a blow down phase provide the way to get a definitive answer regarding the additional fuel fragmentation, before the relocation, associated with the stresses induced by stored energy redistribution.

Now regarding the needs for bundle geometry, the IPSN point of view is the following.

- Bundle geometry is a necessity to get a correct azimuthal temperature field, which is a prerequisite to produce a realistic balloon size.

- The previous PHEBUS LOCA program has demonstrated that the radial interactions between adjacent fuel rods need to be included because they modify the size and shape of the balloons.

- Having in mind that the amount of relocated fuel is associated with the size and the shape of the balloon, it means that realistic data will require bundle geometry.

- Finally, for estimating how much the fuel relocation affects the coolability; it is rather obvious bundle geometry is a necessity.

Then, in a last part, the IPSN paper provides an overview of the feasibility studies engaged for the preparation of such an in-pile program.

An Overview of NRC Research Activities in Probabilistic Risk Analysis

S. Newberry, P. Baranowsky, M. Cunningham
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC, USA

The NRC's research program in probabilistic risk analysis includes a spectrum of activities, from basic research to regulatory applications. This research program includes the following:

1. Development and demonstration of methods which improve existing techniques or fill gaps in the current state of PRA technology.

   Key activities currently underway within this element are developing methods for human reliability analysis, fire risk analysis, and consideration of aging effects in PRAs.

2. Development and demonstration of advanced models and tools for use by the NRC staff and others performing risk assessments.

   For the past several years, the staff has been developing its "Simplified Plant Analysis Risk" (SPAR) models for use in a variety of regulatory applications. In concert with this, the staff has been making improvements to its SAPHIRE computer software to ensure user-friendly interfaces with the SPAR models, as well as with other PRAs available in the SAPHIRE data base.

3. Collecting and assessing plant operational data.

   The staff works closely with the reactor industry to use data from licensee event reports and other sources to assess the reliability of key reactor systems as well as trends in industry performance.

4. Review of risk assessments performed by licensees or the staff.

   The key activities in this element have been the reviews of the IPEEEs that were performed by licensees of each U.S. nuclear power plant in response to an NRC request in Generic Letter 88-20. The reviews have recently been completed, with key results and perspectives documented in NUREG-1724.

5. Application of PRA methods and tools to support the resolution of regulatory issues.

   The staff has been applying PRA methods and tools to assess possible changes to the technical requirements of 10CFR50 (including specific work on changes to 50.44, 50.46, and 50.61), to technical issues facing the staff, including the assessment of steam generator performance during severe accidents, and the development of risk-based performance indicators that objectively measure aspects of reactor licensee performance and can be used in the agency's reactor oversight process.

This page left blank intentionally

Pressurized Thermal Shock Risk Assessment

Donnie Whitehead and Vince Dandini - Sandia National Laboratories
Alan Kolaczkowski - Science Applications International Corp.
Eric Thornsbury, Nathan Siu, and Roy Woods - US Nuclear Regulatory Commission

The U. S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research is developing the technical basis for modifying the Pressurized Thermal Shock (PTS) screening criteria specified in title 10 Part 50.61 of the U. S. Code of Federal Regulations. As part of this effort a probabilistic analysis of the risk posed by PTS in U. S. pressurized water reactors (PWRs) is being performed. This analysis is an update of previous studies that were done in the early to mid 1980's that supported the development of the original (present) version of the rule. The current analysis will incorporate improvements made in probabilistic risk analysis (PRA), thermal-hydraulics (TH), and probabilistic fracture mechanics (PFM) since the original analyses were performed. The overall objective of this effort is to produce realistic risk input to support the re-evaluation of the PTS screening criteria.

For the PRA portions of the analysis, event trees are used to model the event sequences relevant to PTS. Each PTS sequence is then analyzed to estimate the expected time-dependent TH response of the plant. This primarily focuses on the pressure and temperature history of the primary coolant in the downcomer area of the reactor pressure vessel. The TH responses are then used (along with physical data on the reactor pressure vessel) in the PFM analyses to produce a frequency of through-wall cracks in the reactor pressure vessel. Four PWRs (Oconee, Beaver Valley, Calvert Cliffs, and Palisades) have been selected for analysis; their results will be extrapolated to represent the population of U.S. PWRs. The current working schedule calls for the four individual analyses to be finished by the end of the calendar year.

Presently, work on the Oconee and Beaver Valley PRAs is nearing completion. Event sequence results regarding dominant accident initiators will be presented at the conference. Important equipment failures and human actions will also be identified for the conference. Analyses of the Calvert Cliffs and Palisades PRA models are in progress, with results due later this year.

This page left blank intentionally

Circuit Analysis In Fire Risk Assessment

S. Nowlen, F. Wyant, Sandia National Laboratories
N. Siu, H. Woods, USNRC

Experience from actual fire events has shown that fire-induced failures in electrical cables can lead to a range of circuit faulting behaviors. Fire tests have demonstrated similar insights. Possible circuit fault modes include loss of circuit motive power, spurious actuation, false indications, triggering (or inhibiting) of permissive interlocks, and loss of circuit control functions. In the context of fire probabilistic risk assessment (PRA) each of these circuit fault modes may have unique implications. Therefore, in order to more accurately quantify fire risk, it is important that one be able to quantify the conditional probability of a specific fault mode of a selected circuit given (assuming) that a postulated fire has damaged an electrical cable associated with that circuit.

To address this problem, it must be recognized that as cables are heated by a fire, they may display one or more of a number of potential failure modes. The cable failure modes of interest include conductor-to-conductor short circuits, conductor-to-external ground short circuits, and loss of conductor integrity. For some circuits, such as instrument circuits, a substantial loss of conductor insulation resistance without formation of a "hard" (or low impedance) short circuit may also constitute cable failure. Each unique mode of cable failure may have an unique impact on the related circuit. Hence, a thorough assessment of circuit faulting effects will require consideration of a range of potential cable failure modes. PRA quantification will require quantitative estimates of cable failure mode likelihoods.

How each cable failure mode impacts a given circuit is determined through circuit analysis. In general, the identification of potential circuit fault modes for a specific case study requires the consideration of a number of cable failure scenarios. Each scenario involves propagating a particular fire-induced cable failure mode through the associated electrical circuit to determine the impact on circuit function. These analyses may also require consideration of time-dependent effects. The failure of a cable is a dynamic process so the failure mode will likely be subject to transitions over time.

This paper discusses efforts underway to advance the state of fire PRA analysis practice to more fully account for circuit failure modes and effects. The paper describes how circuit analysis fits into the overall context of a fire PRA, presents a framework for incorporating advanced methods of circuit analysis into future PRAs, and discusses circuit faulting insights gained from operating experience (actual fire events). Also discussed are experimental results relevant to the problem. Included is a review of past cable fire tests that provide some indication of the failure mode of electrical cables exposed to fires. In addition, a set of recent fire tests conducted in cooperation with the U.S. commercial nuclear power industry to assess cable failure modes and circuit fault effects is described and discussed. The paper then closes with a discussion of potential applications issues that must be overcome in order to support a practical application of the proposed circuit analysis methods.

This page left blank intentionally

Risk-Based Performance Indicators and the Inspection Process

Patrick Baranowsky, Chief
Operating Experience Risk Analysis Branch
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Status of RBPI Development:

The draft Phase-1 report on risk-based performance indicator (RBPI) development was published in February 2001. Three initiating event frequency indicators were identified under the initiating events cornerstone of safety. Under the mitigating systems cornerstone of safety, thirteen potential RBPIs for BWRs and eighteen potential RBPIs for PWRs were identified. These involved unreliability and unavailability indicators with plant-specific performance thresholds at the train-level for risk-significant safety systems and cross-system performance of key components. The final Phase-1 RBPI development report will be published in November 2001.

Follow-on work in the RBPI development program includes support for a proposed pilot program beginning in 2002 to evaluate potential plant-specific unreliability indicators for the six mitigating systems under the current Reactor Oversight Process (ROP), as well as changes to the current Safety System Unavailability Performance Indicators (SSUPIs). In addition, risk-informed thresholds for the industry trends program will be developed in response to the SRM on SECY-01-0111using insights from the Phase-1 RBPI development efforts.

This page left blank intentionally

EPRI Strategic Action Plan for Risk Technology

John Gaertner, EPRI, Nuclear Strategic Bridge Plan Project Manager
Jack Haugh, EPRI, Manager of Risk and Reliability

EPRI has an Electricity Technology Roadmap which identifies opportunities and threats for innovation over the next 25 years and beyond. There is a Nuclear Sector Strategic Bridge Plan with a clear vision and strategic objectives to achieve Roadmap destinations. A strategic action plan has been developed for Nuclear Risk Technology that focuses near-term R&D on these long-term opportunities. EPRI risk technology R&D can support 11 of 17 nuclear strategic objectives. However, the plan identifies 6 significant barriers to success. Five of these barriers prevent effective use of risk-informed regulations. The plan delineates and critically evaluates all current R&D activities that address these barriers. Most importantly, the plan identifies 5 high priority actions with the greatest potential to break down these barriers. This paper discusses the barriers and high priority actions in detail with emphasis on the impact of these items on risk-informing regulations and operations. The paper also discusses innovative collaboration and funding opportunities to accomplish these actions.

This page left blank intentionally

NOTES

Notes

NOTES

Notes

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CP-0174
2. TITLE AND SUBTITLE Transactions of the Twenty-Ninth Nuclear Safety Research Conference (formerly, the Water Reactor Safety Information Meeting)	3. DATE REPORT PUBLISHED MONTH: October YEAR: 2001 4. FIN OR GRANT NUMBER A3988
5. AUTHOR(S) Conference Papers by various authors; Compiled by Susan Monteleone, BNL	6. TYPE OF REPORT 7. PERIOD COVERED *(Inclusive Dates)* October 22-24, 2001

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as Item 8 above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager; Transactions prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This contains summaries of papers on reactor safety research to be presented at the 29th Nuclear Safety Research Conference (formerly known as the Water Reactor Safety Information Meeting) held at the Marriott Hotel at Metro Center, Washington, DC, October 22-24, 2001. The summaries briefly describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Also included are summaries of invited papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry. The summaries have been compiled in one report to provide a basis for meaningful discussion of information exchange during the course of the conference, and are given in the order of their presentation on each day of the conference.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
reactor safety research nuclear safety research	unlimited
	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified
	15. NUMBER OF PAGES
	16. PRICE

This form was electronically produced by Elite Federal Forms, Inc.

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300